CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9902110166 DOC.DATE: 99/02/01 NOTARIZED: NO DOCKET #
FACIL:50-287 Oconee Nuclear Station, Unit 3, Duke Power Co. 05000287
AUTH.NAME AUTHOR AFFILIATION
BURCHFIELD,J.E. Duke Power Co.
MCCOLLUM,W.R. Duke Power Co.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 98-004-00:on 981231,reactor trip occurred during CRD circuit breaker trip test.Caused by broken wire.Repaired broken wire,enhanced maint work practices & enhanced test methods.With 990201 ltr.

DISTRIBUTION CODE: IE22T COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 10
TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD2-2 PD	1	1	LABARGE,D	1	1
INTERNAL:	ACRS	1	1	AEOD/SPD/RAB	2	2
	AEOD/SPD/RRAB	1	1	FILE CENTER	1	1
	NRR/DE/ECGB	1	1	NRR/DE/EELB	1	1
	NRR/DE/EMEB	1	1	NRR/DRCH/HICB	1	1
	NRR/DRCH/HOHB	1	1	NRR/DRCH/HQMB	1	1
	NRR/DRPM/PECB	1	1	NRR/DSSA/SPLB	1	1
	RES/DET/EIB	1	1	RGN2 FILE 01	1	1
EXTERNAL:	L ST LOBBY WARD	1	1	LITCO BRYCE,J H	1	1
	NOAC POORE,W.	1	1	NOAC QUEENER,DS	1	1
	NRC PDR	1	1	NUDOCS FULL TXT	1	1

C A T E G O R Y 1 D O C U M E N T

NOTE TO ALL "RIDS" RECIPIENTS:
PLEASE HELP US TO REDUCE WASTE. TO HAVE YOUR NAME OR ORGANIZATION REMOVED FROM DISTRIBUTION LISTS OR REDUCE THE NUMBER OF COPIES RECEIVED BY YOU OR YOUR ORGANIZATION, CONTACT THE DOCUMENT CONTROL DESK (DCD) ON EXTENSION 415-2083

FULL TEXT CONVERSION REQUIRED
TOTAL NUMBER OF COPIES REQUIRED: LTTR 23 ENCL 23

Duke Energy

W. R. McCollum, Jr.
Vice President

Duke Energy Corporation
Oconee Nuclear Station
P.O. Box 1439
Seneca, SC 29679
(864) 885-3107 OFFICE
(864) 885-3564 FAX

February 1, 1999

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287
Licensee Event Report 287/98-04, Revision 0
Problem Investigation Process No.: 3-098-6101

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a) (1) and (d), attached is Licensee Event Report 287/98-04, concerning a Unit 3 Reactor Trip.

This report is being submitted in accordance with 10 CFR 50.73 (a) (2) (iv). This event is considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

for WRM

W. R. McCollum, Jr.

Attachment

cc: Mr. Luis A. Reyes
Administrator, Region II
U.S. Nuclear Regulatory Commission
61 Forsyth Street, S. W., Suite 23T85
Atlanta, GA 30303

Mr. D. E. LaBarge
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

INPO Records Center
700 Galleria Parkway, NW
Atlanta, GA 30339-5957

Mr. M. A. Scott
NRC Resident Inspector
Oconee Nuclear Station

NRC FORM 366
(4-95)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMD NO. 3150-0104
EXPIRES: 04/30/98
ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Oconee Nuclear Station, Unit 3	05000 287	1 of 8

TITLE (4)
Broken Wire Causes Reactor Trip During Control Rod Drive Breaker Test

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER(S)
										05000
12	31	98	1998	- 04 -	0	02	01	99		05000

OPERATING MODE (9)	N
POWER LEVEL (10)	100 %

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR (*Check one or more of the following*) (11)

	20.402(b)		20.405(c)	X	50.73(a)(2)(iv)		73.71(b)
	20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
	20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER (*Specify in Abstract below and in Text, NRC Form 366A*)
	20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		
	20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
	20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (AREA CODE)	
J.E. Burchfield, Regulatory Compliance Manager	(864)	885-3292

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
B	AA	RLY	D150	Y					

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (*f yes, complete EXPECTED SUBMISSION DATE*)	X NO				

ABSTRACT (*Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines*) (16)

On December 31, 1998, Unit 3 experienced a reactor trip from 100% Full Power during a Control Rod Drive (CRD) circuit breaker (BKR) trip test. When the test technicians tripped the first BKR per the procedure, the CRD Channel 'B' Trip Confirm Relay actuated unexpectedly. This actuation caused a Generator Lockout, followed immediately by a Main Turbine Trip and Anticipatory Reactor Trip. Unit response was routine and the operators stabilized the unit. A broken wire de-energized an auxiliary relay, which simulated two of the four DC CRD BKRs being open. When the AC BKR under test was opened, the Trip Confirm Relay logic caused the Generator Lockout. Investigation concluded that the wire broke during maintenance on an adjacent circuit during the most recent refueling outage. The trip occurred on the first test at power after the outage. The root cause was a defect dated from manufacture, with a contributing preventative maintenance weakness that allowed a loose fuse holder to cause movement of the wire. Corrective actions include repair of the broken wire, enhancement of maintenance work practices, and enhancement of test methods to allow verification of relay status prior to testing a BKR. The health and safety of the public was not compromised by this event.

EVALUATION:

Background

The Reactor Protective System (RPS) [EIIS:JC] is a safety related system which monitors parameters related to the safe operation of the plant. The RPS provides a two-of-four logic for tripping the reactor in response to unit/system conditions that require a unit trip. This is accomplished via the reactor trip module relays [EIIS:RLY] which de-energize the Control Rod Drive (CRD) System AC and DC circuit breakers (BKRs) causing all control rods to drop.

The CRD System operates the reactor control rods. If power is interrupted to the CRD motors, the control rods are allowed to fall into the core, shutting down the reaction and tripping the unit. The CRD system contains two AC BKRs (designated 'AC Breaker Unit 10' and 'AC Breaker Unit 11') and four DC circuit BKRs (designated CB 1 through 4). The AC BKRs control all the 3 phase primary power to the rod drives. The DC BKRs control the DC power to rod groups 1 through 4, the gating power to rod groups 5 through 8, and the auxiliary power supplies. The control rod drive BKR combinations that initiate a reactor trip are:

1) open both AC BKS
2) open all 4 DC BKRs
3) open AC BKR 10 with DC BKRs CB 3 and 4
4) open AC BKR 11 with DC BKRs CB 1 and 2

The status of the AC and DC BKRs is monitored by auxiliary relays (designated K1 through K4) which provide logic to activate an additional relay 'K5'. K5 provides a Reactor Trip Confirm, Channel B, signal which generates a Generator Lockout. The Generator Lockout causes a Main Turbine Trip, and trips other secondary side components. The Main Turbine Trip results in an Anticipatory Reactor Trip to the RPS. The Anticipatory Reactor Trip may be bypassed at low power levels. There is a redundant logic that provides a Reactor Trip Confirm Channel A signal which activates a Generator Back-up Lockout and other actions, again leading to an Anticipatory Reactor Trip.

Description of Event

On December 31, 1998, Unit 3 was operating at 100% Full Power (FP). A periodic surveillance test was in progress on the Control Rod Drive (CRD) system. The intent of the test was to verify operability of the CRD circuit breakers (BKRs) by tripping them one at a time, so that the 2 of 4 trip logic would not be actuated.

At approximately 1415 hours, Reactor Protective System (RPS) Channel A was placed in Manual Bypass for the test. At 1435 hours, the Instrument and Control (I&C) technicians performing the test initiated a shunt trip of AC BKR CB 10 in accordance with their procedure. They and the Control Room Operators noted an immediate reactor trip.

Other than the RPS trip signals which initiated the trip, there were no emergency actuations or abnormal responses associated with this trip. The Emergency Power Switching Logic transferred power from the normal source (the unit generator) to the startup source (switchyard) as designed. Operator response was immediate and appropriate. RPS Channels B, C, and D all tripped. Channel A did not trip since it was in Manual Bypass for the test. The operators verified that all control rods fully inserted and began immediate post-trip manual actions. The Transient Monitor Computer was out of service at the time of the trip but data obtained from the Operator Aid Computer was adequate for the post-trip analysis.

As normally occurs in the first minute following a trip, the Reactor Coolant System (RCS) [EIIS:AB] inventory shrank as it cooled from 579F (normal average RCS temperature) to approximately 550F. Therefore, Pressurizer level, initially at 219 inches, dropped to 41 inches. To compensate for RCS inventory shrinkage, an operator manually started a second High Pressure Injection (HPI) [EIIS:CB] pump and opened valve 3HP-26 (HPI TO LOOP A REACTOR INLET VALVE) to increase normal make-up to the RCS. After three minutes, Pressurizer level was restored to approximately 130 inches and the operator secured the second HPI pump and closed 3HP-26. RCS pressure was 2130 psig prior to the trip and decreased to 1751 psig in the initial transient before increasing back to 2130 psig.

Steam Generator (secondary) pressure was 895 psig prior to the trip and increased to 1092 psig immediately after the trip. Main Steam Relief Valves opened as expected during the pressure peak. Steam pressure dropped to approximately 980 psig momentarily before stabilizing at 1010 psig. While the secondary side was at this pressure, the RCS temperature increased to 562F. Approximately fifteen minutes after the trip the operators took action, as directed in the procedure, to reduce steam pressure to 980 psig to ensure that the Main Steam Relief Valves had reseated. After this action, the RCS temperature reduced to 555F (post-trip temperature setpoint).

Steam generator inventory reduced normally from the normal operating level to 25 inches as indicated on the Start-up Level instruments. Main feedwater pumps ran back as necessary and continued to supply feedwater. Emergency feedwater was not needed and did not actuate.

A troubleshooting work plan was developed and implemented per procedure. A visual inspection was performed which revealed a broken wire on fuse 'F4' in CRD Trip Confirmed Channel 'B' circuitry. This broken wire prevented relay 'K4' from energizing, which caused contacts associated with DC BKRs CB 3 and CB 4 to remain in the tripped state. In effect, the DC BKRs CB 3 and CB 4 logic indicated a trip at all times to the Channel 'B' Trip Confirmed relay 'K5'. However, there are no alarms or indicating lights showing the status of the 'K4' relay. When the CB-10 BKR was opened for the test, the logic was satisfied for the trip confirm circuit. The Channel 'B' Trip Confirmed relay generates a Generator Lockout, which results in a Main Turbine Trip and an Anticipatory Reactor Trip.

The investigation determined the cause of the broken wire in the CRD logic assembly.

The investigators concluded that the wire was broken due to metal fatigue. They found that the multi-strand wire had been scored during insulation removal as part of connection during manufacturing, in the early 1970's. When the lead was soldered, solder was wicked into the scored area. This combination created a rigid assembly and a stress point.

In addition, the fuse holders are of a type that allows "keying" to prevent rotation; however, the mounting holes were not punched with the proper key shape. This allowed relatively free rotation of the fuse holder as the mounting nut loosened over time. The result was that the 'F4' load wire was flexed at the stress point during fuse checking/replacement over many years.

This Unit 3 reactor trip occurred during the first CRD Trip Test following startup after the outage. The unit did not trip during tests prior to the outage. A review of work activities during the 3EOC17 refueling outage revealed that between October 20, 1998 and November 30, 1998 the 'F5' fuse holder was replaced. The 'F5' fuse holder is adjacent to the 'F4' holder, which had the broken wire.

A review of computer records for the Sequence of Events Recorder (SER) indicates that the 'K5' relay was inappropriately actuated during outage testing on November 30, 1998, following the replacement of the 'F5' fuse holder. Therefore the investigation team concluded that the wire was weakened over an extended period of time and was broken inadvertently during that maintenance activity.

The SER printout clearly shows multiple alarms for 'Reactor Trip Confirm B' during tests. The SER printout was used to verify that the various BKRs had tripped during the test. However, neither the test procedure nor training required the technicians to observe SER points other than those specifically expected during the test. Also, the various components tripped by 'K5' relay actuation, including the turbine and RPS, were already in the tripped state and did not react. Therefore, the 'K5' actuation was not recognized as abnormal at the time.

The unit was returned to criticality at 0153 hours on January 2, 1999 and reached 100 %FP at 0300 hours on January 3, 1999.

CONCLUSIONS

The root cause of this reactor trip was a latent defect, (scoring of the conductor strands when stripping insulation) which dated from manufacture of the circuit and created a stress point in the wire. The investigation team concluded that the wire was weakened due to the existence of the stress point and was broken inadvertently during a maintenance activity.

Contributing to this was a weakness in the preventative maintenance program in that it did not require the fuse holder mounting nut to be retightened, if loose, when checking or replacing fuses. The loose mounting nut allowed the fuse holder to move, and caused the wire to flex at the stress point during periodic fuse checking and/or replacement. The investigation concluded that this weakened the wire over time such that it broke during routine maintenance on an adjacent fuse holder.

A review for similar events within the previous two years did not identify any additional instances of broken wires associated with scoring of the conductor strands when stripping insulation or poor soldered connections. Inspection of the wiring on the other relays in the CRD Trip Confirm Logic did not reveal any additional deficiencies. Therefore, this is considered an isolated occurrence and no additional corrective actions to address this root cause are considered necessary at this time.

The contributing weakness in the preventive maintenance program is associated with an ongoing issue of material condition and work standards. Planned corrective action 1 will address this issue.

The review for similar events within the previous two years identified that, on March 30, 1997, Unit 3 experienced a trip during CRD BKR testing, reported in LER 50-287/97-01. That trip was due to a blown fuse on CRD relay 'K3', associated with the 'Reactor Trip Confirm A' relay. In that case the cause was determined to be a screw on a wiring plug connection that had apparently pinched the insulation. Over a period of years the insulation was stressed to the point that the wire contacted the screw and caused the fuse to blow. That investigation identified that the only indications of circuit malfunction associated with the K1 through K5 relays are blown fuse indicators on the fuse holders. In the 1997 event, the 'F3' fuse was blown, but the blown fuse indicator was burned out. Corrective actions included replacing the failed fuse and indicator, and revising the appropriate procedures to inspect the fuse indicators for blown fuses before initiating the BKR test. The intent of these procedure revisions was to assure that the associated relays were

APPROVED OMB NO. 3150-0104
EXPIRES:4/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503

energized and in the proper state to allow testing without a unit trip. This corrective action was ineffective to prevent the December 31, 1998 trip because it did not address all potential failure modes of the circuits. Specifically, it would not identify an open circuit de-energized for any reason other than a blown fuse. Planned corrective action 2 addresses this issue.

The failure of the circuit is reportable under the Equipment Performance and Information Exchange (EPIX) program. The affected component was the AC Breaker B Redundant Trip Confirm Assembly, a portion of the Control Rod Drive System, manufactured by Diamond Power Specialty Corp.

There were no releases of radioactive materials, or personnel injuries or exposures associated with this event.

CORRECTIVE ACTION:

Immediate:

1. The operators established stable hot shutdown conditions.

Subsequent:

1. Instrument and Control (I&C) technicians discovered the broken wire and re-soldered the connection.

2. I&C technicians inspected the other fuses for similar problems in both Channels A and B on Unit 3. Also, various CRD auxiliary relays were tested for proper operation.

3. The fuse holders for the Channel A and Channel B Trip Confirm relays on Unit 3 were inspected and tightened as necessary to prevent rotation.

4. I&C technicians performed post-maintenance testing by running the CRD Breaker Trip Test procedure and verifying proper indications on the Sequence of Events Recorder.

Planned:

1. I&E Maintenance will prepare a "lessons learned" communication to craft technicians to enhance awareness of the contribution of minor housekeeping and material condition defects to equipment failures. This communication will address the issue from this event relative to the loose fuse holder mounting nuts. The intent of the communication is to reinforce Management's expectation that observation and correction of such deficiencies is part of good work standards.

2. Engineering will evaluate methods to positively verify position/status of auxiliary relays K1 through K4. (E.G. voltage tests, coil-monitoring LEDs, position indicting lights). After determining the appropriate method, appropriate procedures associated with the breaker tests will implement verification of relay status prior to performing CRD breaker tests.

Planned corrective actions 1 and 2 are considered to be NRC Commitment Items. These actions are the only NRC Commitment items contained in this LER.

SAFETY ANALYSIS:

This unit trip was due to a broken wire. The broken wire caused the associated circuits to go to their designed fail safe position, which effectively reduced the 2-of-4 Reactor Protective System logic to 1-of-3. A unit trip is the desired safe response in the event of such failures. Therefore, the plant performed as designed to enter a safe state.

Post-reactor trip response, as discussed in the Event Description section of this report, was within acceptable limits as defined by the Babcock and Wilcox Owners Group Transient Assessment Program.

Therefore, the health and safety of the public was not affected by this event.